ACCESSION #: 9605220343
LICENSEE EVENT REPORT (LER)

FACILITY NAME: WATTS BAR NUCLEAR PLANT - UNIT 1 PAGE: 1 OF 8

DOCKET NUMBER: 05000390

TITLE: Manual Turbine Trip/Automatic Reactor Trip
EVENT DATE: 04/16/96 LER #: 96-014-00 REPORT DATE: 05/14/96

OTHER FACILITIES INVOLVED: DOCKET NO: 05000

OPERATING MODE: 1 POWER LEVEL: 083

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION:
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
NAME: Rickey Stockton, Compliance Licensing
Engineer TELEPHONE: (423) 365-1818

COMPONENT FAILURE DESCRIPTION:
CAUSE: B SYSTEM: JJ COMPONENT: EI MANUFACTURER: W351
REPORTABLE NPRDS: NO

SUPPLEMENTAL REPORT EXPECTED: NO

ABSTRACT:

On April 16, 1996, at 1034 hours EDT, Unit 1 was operating in Mode 1 at 83 percent rated thermal power (RTP) when the governor valves (GV) to the high pressure turbine automatically closed resulting in loss of turbine load from 1000 MWe. In response, the licensed unit operator manually tripped the turbine which automatically tripped the reactor since RTP was greater than 50 percent. All control rods inserted normally and the unit was stabilized in Mode 3. In accordance with the plant design, the automatic reactor trip was followed by a main feedwater isolation resulting in a start of all auxiliary feedwater pumps.

The subsequent investigation determined that the cause was due to the interface of the automatic self-check circuit of the Anticipated Transient Without Scram (ATWS) Mitigation System Actuation Circuit (AMSAC) system with that of the overspeed protection controller (OPC) of the main turbine. Corrective actions included the disabling of the AMSAC self-test circuitry.

END OF ABSTRACT

I. PLANT CONDITIONS

Watts Bar Nuclear Plant Unit 1 was operating in Mode 1 at 83 percent RTP producing approximately 1000 MWe prior to the event.

II. DESCRIPTION OF EVENT

A. Event

On April 16, 1996, at 1034 hours EDT, the GV (Energy Industry Identification System (EIIS) code 65/FCV) (1-FCV-001-0062, -0065, -0068, and -0071) to the high pressure turbine (EIIS code TRB) automatically closed resulting in loss of turbine load from 1000 MWe. In response, the licensed unit operator manually tripped the turbine (EIIS code TRB) which automatically tripped the reactor (EIIS code RCT) since RTP was greater than 50 percent. All control rods inserted normally and the unit was stabilized in Mode 3. The automatic reactor trip was followed by a main feedwater isolation resulting in a start of all auxiliary feedwater pumps (EIIS code BA/P).

Subsequent investigation determined that, during the automatic self-check of the AMSAC system which occurs every 13 days and 11 hours, the OPC solenoid (EIIS code FSV), FSV-047-0026B, was actuated. When this solenoid energizes, electrohydraulic control (EHC) fluid is released from the GV emergency fluid header ultimately resulting in closure of the governor valves.

B. Inoperable Structures, Components, or Systems that Contributed to the Event

There were no inoperable structures, components, or systems that contributed to the event.

C. Dates and Approximate Times of Major Occurrences

TIME EVENT

10:33:54 The loss of load steam dump interlock alarm (EIIS code ALM) actuated in the Main Control Room (MCR). GV and intercept valve (IV) closures were initiated

by overspeed protection circuit initiation due to an
applied voltage from an AMSAC self-test circuit.
Indications were decrease in main turbine impulse
pressure and reheat pressure, and increase in T sub
avg due to loss of load.

10:33:55 All steam dump valves (EIIS code FCV) 0 -FCV-1 -1 03,
-107, -111, -104, -108, 112, -105, -109, -113, -106,
-110, and -114) automatically opened.

10:33:56 AMSAC abnormal alarm occurred in 2 seconds following
test of OPC initiation relay (EIIS code RLY)
(RS31-4A) from low field device impedance detection.

10:34:10 The unit operator placed the control board rod
control in the automatic position and the control
rods began to insert.

10:34:19 Steam generator (SG) (EIIS code SG) number 1 power
operated relief valve (PORV) (EIIS PCV) lifted.
Within seconds the remaining 3 PORVs lifted.

10:34:42 The unit operator manually tripped the turbine
followed by an automatic reactor trip which occurred
because RTP was greater than 50 percent. The
automatic reactor trip was followed by a main
feedwater isolation resulting in a start of all
auxiliary feedwater pumps.

D. Other Systems or Secondary Functions Affected

Refer to Section IV.

E. Method of Discovery

The condition was immediately identified by control room indication.

F. Operator Actions

The operating crew manually tripped the turbine when the generator
output went from 1,000 to 0 MWe and entered Abnormal Operating
Instruction (AO1) -17, "Turbine Trip"

After starting normally, the turbine driven auxiliary feedwater pump was removed from service to minimize cooldown.

The standby main feedwater pump (EIIS code SJ/P) was started.

G. Automatic and Manual Safety System Response

As a result of the manual turbine trip above 50 percent reactor power, an automatic reactor trip with control rod insertion occurred as expected. As designed, the reactor trip was followed by a main feedwater isolation resulting in a start of all auxiliary feedwater pumps. The turbine driven auxiliary feedwater pump was eventually removed from service to minimize cooldown.

III. CAUSE OF EVENT

The root cause of the April 16, 1996, turbine trip and a similar one which occurred on March 18, 1996, has been determined to be an improper design interface between the self-testing circuit of the AMSAC system and the main turbine OPC. The AMSAC system conducts a self-test on a frequency of 13 days and 11 hours. During self-test, approximately 39 volts AC is applied to the

OPC solenoid, 1-FSV-47-26B, causing EHC control valve emergency fluid header to be depressurized. This closes the GV and IVs for about 2 seconds. GVs did not reclose as the servo valves for the GV actuators were demanding EH fluid to open the GVs while the dump valves were unseated. Adequate differential pressure (DP) across the disc dump valve could not be developed to seat the disc dump valve.

The timing of this self-test was such that the March 18, 1996, and the April 16, 1996, events were on timed test days. Deviations in exact 13 days and 11-hour intervals are due to the need to manually reset the AMSAC self-test logic before it will restart the self-test routine. No turbine trip or auxiliary feedwater (AFW) pump initiations from AMSAC were observed as the self-test stops when a deviation is sensed. OPC initiation relays are checked before turbine trip and AFW initiation relays. Thus, OPC initiation relay deviations block the other relay testing. This output relay testing occurs near the end of the test sequence.

An AMSAC alarm occurred on April 2, 1996, at 0950 hours EST when reactor power was less than 18 percent with the turbine generator in service. The last AMSAC alarm reset was on April 2, 1996, at 2307 hours EST (13 days and 11 hours later) which would cause a self-test on April 16, 1996, at about 0900 correcting for daylight savings time.

A contributing factor to this event was the inadequate determination of the AMSAC system abnormal alarm identified during the March 18, 1996 turbine trip event. This incorrect assessment of the cause of the alarm under the previous investigation was due to the pre-existing deviation between main turbine impulse pressure inputs which had generated previous spurious alarms and the rapid change in impulse pressure upon the event initiation.

IV. ANALYSIS OF EVENT - ASSESSMENT OF SAFETY CONSEQUENCES

A. Evaluation of Plant Systems/Components

Reactor Coolant System/Reactor Core Response

The plant responded as expected to the rapid decrease in turbine power and subsequent turbine and reactor trips. Rod control was in manual and the operator at the controls (OAC) placed control rods in automatic in response to the load rejection. RCS temperature followed the reduction in power once the turbine and reactor trips were initiated. One rod bottom light (M did not light. The bulb was replaced and the rod verified to be inserted.

Condensate/Feedwater

Prior to the event, the condensate/feedwater system (EllS codes SG/SJ) was aligned for full power operation. Three hotwell pumps (HIS code SD/P) and three condensate booster pumps (EllS code SD/P) were in service supplying suction to main feedwater pumps (EllS SJ/P) 1A and 1B. Two number 7 and three number 3 heater drain pumps (EIIS code SM/P) were in service pumping forward. Main feedwater pumps 1A and 1B were in service and in automatic. The main feedwater regulating valves were in automatic.

At the initiation of the event, steam flow decreased and steam pressure increased due to rapid closing of the main turbine governor valves. Subsequently, steam dump valves (EIIS code FCV) and atmospheric dump valves (EIIS code PCV) opened. Steam generator (SG) water levels began to slowly decrease in conjunction with the steam pressure increase and the steam flow decrease. The main feedwater regulating valves operated as expected in automatic control during this portion of the transient.

About 20 seconds after manual initiation of turbine trip and subsequent reactor trip, a main feedwater isolation (MFWI) occurred as expected. The MFW isolation valves (EIIS code FCV) closed followed by an automatic trip of both turbine driven main feedwater pumps (TDMFP's). This resulted in the auto initiation of all auxiliary feedwater (AFW) pumps.

Electro-Hydraulic (EH) Fluid System

About 9 seconds after initiation of the event, an EH fluid low pressure condition alarmed as expected due to GV opening. OPC initiation removes control valve emergency fluid from the 4 GVs and 6 IVs to the drain tank, significantly increasing pump flow demand and dropping pressure. The standby electro-hydraulic controller (EHC) pump started about 15 seconds into the event on low pressure. This EH fluid system transient was more severe than the March 18, 1996 event. The generator loss of load as the OPC signal to close GV's and IV's was not present, and a much greater flow demand existed with disc dumps open and normal 83 percent power GV position demand present simultaneously.

Steam Dumps to Condenser

Steam dumps tripped open as a result of GVs and IVs closing on the main turbine causing a total loss of turbine generator load. This condition resulted in immediate decrease in turbine impulse pressure. The anticipatory circuit for T sub ave confirmed the high rate of change and tripped open the dump valves. All four groups of steam dump valves tripped opened as expected. All steam dump valves closed 1 minute and 25 seconds after event initiation.

SG Atmospheric Dump Valves

Main steam line pressures increased to about 1115 psig. All atmospheric dump valves were open at 31 seconds after initiation of the event. All 4 atmospheric dump valves closed 1 minute and 12 seconds after event initiation.

B. Evaluation of Personnel Performance

In response to the rapid drop in generator output, the operating crew manually tripped the turbine which initiated a reactor trip. Operators responded by using the appropriate Emergency Operating and Abnormal Operating Instructions to place the plant in Mode 3.

Feedwater control was further affected by the 'shrink & swell' of SG levels associated with the steam flow perturbations. The operating crew took manual control of SG level control

valves and AFW initiation to control cool down. The turbine driven auxiliary feedwater pump (TDAFWP) flow was reduced and the pump subsequently tripped to control cool down. The motor driven auxiliary feedwater pump (MDAFWP) level control valves were then throttled to further control cool down.

Reactor trip logic may have been challenged unnecessarily due to operator action to trip the turbine rather than the reactor directly at the existing power level.

C. Safety Significance

The loss of load/turbine trip as described in the FSAR has initial plant conditions of 102% RTP with the following assumptions: The steam dumps and the SG atmospheric dump valves are not assumed to be functioning. Reactor controls are assumed to be in manual and the plant relies on SG safety valves for pressure relief. Main feedwater flow is assumed to be lost at the time of loss of load. No credit is taken for the reactor trip or turbine trip. The plant receives a reactor trip from high pressurizer pressure.

This plant transient follows the FSAR transient very well and the response of plant equipment was well within the design basis. The plant transient relied on steam dumps and SG atmospheric dump valves to remove the heat load whereas the FSAR transient conservatively does not utilize this benefit. The plant transient was terminated by a manual turbine trip which automatically initiated a reactor trip. Main feedwater flow was not lost until 66 seconds following the loss of load. The plant responses and conditions were consistent with and are considered bounded by the FSAR transient.

In conclusion, since the steam dumps to the condenser and the atmospheric relief valves were available to reduce the likelihood of tripping the reactor on a high pressurizer pressure signal, and thus preclude opening of the pressurizer safety valves, the reactor was automatically tripped on turbine trip above 50% power. The plant responded well within the design basis as described in the FSAR.

V. CORRECTIVE ACTIONS

A. Immediate Corrective Actions

The immediate corrective actions for this event are described under the operator actions section of this report.

B. Corrective Actions to Prevent Recurrence

1. TVA has implemented Design Change Notice (DCN) W-3881 O-A for the removal of the AMSAC trip in OPC, to disable the AMSAC output self-test logic, to disable CIV fast closure circuitry in EHC, and to remove the low oil pressure switch 1-PS-47-76 for the OPC circuit.

2. TVA has implemented DCN W-38809-A for the removal of the parallel solid state (TRIAC) relays in the EHC identified during the investigation as a no longer needed portion of the circuit.

3. TVA has provided lessons learned training to technical support engineers regarding the inadequate evaluation of cause of AMSAC system abnormal alarms prior to this event.

VI. ADDITIONAL INFORMATION

A. Failed Components

1. Safety Train Inoperability

There were no failures that rendered a train or safety system inoperable.

2. Component/System Failure Information

a. Method of Discovery of Each Component or System

Failure:

Loss of Turbine Load - The root cause for this failure was determined by thorough analysis of the event data and through inspection of the equipment. The investigation team used Kepner-Tregoe (KT) analysis techniques to determine the root cause of the event.

b. Failure Mode, Mechanism, and Effect of Each Failed Component:

The failure of the loss of turbine load was due to a design deficiency. The effect of the AMSAC self-test circuitry on the main turbine OPC circuitry was not fully understood prior to installation.

c. Root Cause of Failure:

Design error - See the cause section of this report.

d. For Failed Components With Multiple Functions, List of Systems or Secondary Functions Affected:

See description of event for functions affected by the loss of turbine load as a result of the AMSAC deficiency.

e. Manufacturer and Model Number of Each Failed Component:

Atomic Energy of Canada Limited (AECL) - AMSAC

B. Previous Similar Events

There have been no similar events in the AMSAC system previously reported under 10CFR50.72 or 10CFR50.73. However, a similar turbine trip occurred on March 1 8, 1 996, which was determined not to be reportable since the event did not result in a reactor trip or an actuation of an engineered safety feature.

VII. COMMITMENTS

The actions taken in response to this event have been completed and are tabulated in Section V, Corrective Actions.

ATTACHMENT TO 9605220343 PAGE 1 OF 3

TVA

Tennessee Valley Authority, Post Office Box 2000, Spring City, Tennessee 37381-2000

MAY 16 1996

John A. Scalice

Site Vice President, Watts Bar Nuclear Plant

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555

Gentlemen:

In the Matter of) Docket Number 50-390
Tennessee Valley Authority)

WATTS BAR NUCLEAR PLANT (WBN) - UNIT 1 FACILITY OPERATING LICENSE NPF-90
- LICENSEE EVENT REPORT (LER) 50-390/96014 - MANUAL TURBINE TRIP/AUTOMATIC REACTOR TRIP DUE TO ANTICIPATED TRANSIENT WITHOUT SCRAM
MITIGATION SYSTEM ACTUATION CIRCUITRY (AMSAC) DEFICIENCY

The purpose of this letter is to provide the subject LER. The enclosed report provides details regarding the subject trip which occurred on April 16, 1996. Submittal of this report is in accordance with 10 CFR 50.73(a)(2)(iv).

If you should have any questions, please contact P. L. Pace at (423) 365-1824.

Sincerely,

J. A. Scalice

enclosure
cc: See page 2

ATTACHMENT TO 9605220343 PAGE 2 OF 3

U. S. Nuclear Regulatory Commission
Page 2

MAY 16 1996

cc (Enclosure):
NRC Resident Inspector
Watts Bar Nuclear Plant
1260 Nuclear Plant Road
Spring City, Tennessee 37381

Mr. Robert E. Martin, Senior Project Manager
U.S. Nuclear Regulatory Commission
One White Flint North
11555 Rockville Pike
Rockville, Maryland 20852

U.S. Nuclear Regulatory Commission
Region II
101 Marietta Street, NW, Suite 2900
Atlanta, Georgia 30323

ATTACHMENT TO 9605220343 PAGE 3 OF 3

ENCLOSURE

*** END OF DOCUMENT ***